ACCESSION #: 9202120181
LICENSEE EVENT REPORT (LER)

FACILITY NAME: COMANCHE PEAK - UNIT 1 PAGE: 1 OF 10

DOCKET NUMBER: 05000445

TITLE: PERSONNEL ERROR WHILE TAKING MANUAL CONTROL OF THE GENERATOR
PRIMARY WATER SYSTEM LEADS TO A REACTOR TRIP
EVENT DATE: 01/08/92 LER #: 92-001-00 REPORT DATE: 02/07/92

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: D.E. BUSCHBAUM, COMPLIANCE TELEPHONE: (817) 897-5851
SUPERVISOR

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On January 8, 1992, the Reactor Operator was attempting to reestablish the required differential temperature between the main generator primary water system and the hydrogen gas system by taking manual control of primary water flow. While in manual control, primary water temperature started to rise and could not be checked, which resulted in a turbine trip followed by a reactor trip occurred due to high primary water temperature.

Root causes were determined to be failure to understand the potential consequences of controlling primary water flow in manual and failure of the Shift Supervisor to adequately monitor the evolution. Contributing factors were failure to use available procedures, lack of specific information in procedures regarding this risk, and the malfunction of the primary water high temperature alarm. Corrective actions include

counselling the specific individuals, intensive training for the crew involved, enhancing requalification training, revising procedures, repairing the alarm, and establishing a task team to evaluate the technical information available to operate and maintain the main generator.

END OF ABSTRACT

1. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS) (EIIS:(JC)).

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On January 8, 1992, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, with reactor power at 100 percent (%).

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

There were no inoperable structures, systems or components that contributed to the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE
TIMES

At 1900, January 8, 1992, the Balance of Plant (BOP) Reactor Operator (RO)(utility, licensed) and trainee (utility, non-licensed) were taking logs on the main generator (EIIS:(TB)) and noticed that the differential temperature (Delta T) between generator hydrogen (EIIS:(TK)) temperature and generator primary water (EIIS:(TJ)) temperature was less than (
that primary water temperature was to be maintained 10 degrees F greater than (>) generator cold hydrogen temperature at all times. The Shift Supervisor (utility, licensed) recommended taking manual control of a temperature control valve to return

Delta T to specification. The intent was to take manual action to correct Delta T and then allow the control system to maintain Delta T >10 degrees F. It was then discussed whether to take manual control of the hydrogen temperature control valve (EIIS:(TCV)(TK)) and lower generator cold hydrogen temperature or to take manual control of the primary water temperature control valve (EIIS:(TCV)(TJ)) and raise primary water temperature. In consideration of a main generator hydrogen pressure drop test that was in progress, the BOP RO decided to take manual control of the primary water temperature controller (EIIS:(TC)(TJ)).

At 1930 the trainee, under the direct guidance of the BOP RO, took manual control of the primary water temperature controller. Primary water temperature was raised from 116 degrees F to 120 degrees F which restored Degrees T to 13 degrees F, within specification.

At 2130, after additional monitoring, the generator primary water/hydrogen gas Delta T was again noted to be less than 10 degrees F. The problem was again discussed among the Control Room staff and the Shift Supervisor recommended sending an Auxiliary Operator (AO)(utility, non-licensed) to ensure that the primary water temperature control valve was not sticking or binding. This recommendation was an effort to determine if the Delta T problem was a result of a problem with the controller or the valve.

At 2145, with an AO at the primary water temperature control valve, the BOP RO took manual control of the valve and closed it from its initial demand position of approximately 10-15%. When the AO reported that the valve was fully closed, the BOP RO opened the valve to the demand position of approximately 50% while the AO checked for proper valve operation. The AO reported that the valve appeared to be moving freely. During this exercise, primary water temperature increased to approximately 120 degrees F when the valve was closed and dropped to 110 degrees F after the valve was opened. The valve was then returned to the initial position of 10-15% open and the controller was returned to auto when primary water temperature had risen to approximately 116-118 degrees F. The primary water temperature, however, continued to increase above 120 degrees F and the controller, in auto, did not appear to be opening the valve fast enough to stabilize temperature. The

BOP RO again took manual control, opened the valve partially and the temperature appeared to stabilize at about 125 degrees F. After two minutes, however, temperature started to rise again and the BOP RO increased valve demand to 35% while temperature increased to 130 degrees F. The Shift Supervisor reminded the BOP RO that maximum allowed primary water temperature was 131 degrees F and the BOP RO increased valve demand position to 40-45%.

At 2200 the Control Room received an automatic turbine/reactor trip due to a generator primary water temperature of 140 degrees F. At the approximate time of the trip the primary water temperature control valve was at a demand position of 40-45% and primary water temperature indicated approximately 135 Degrees F. The generator primary water high temperature alarm (EIIS:(TA)(TJ)) did not come in during operation of the primary water temperature control valve or prior to the turbine/reactor trip.

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Following the event, troubleshooting was performed on the primary water temperature control valve and associated control loop. The purpose was to determine why the control loop did not seem to respond properly in auto and why the generator primary water high temperature alarm did not come in prior to the trip. No problems were found with the primary water temperature control valve that would have impaired proper operation of the valve. The control loop, however, was determined to have several problems:

o primary water flow temperature switch had failed and would not have actuated the alarm under these conditions, and

o three other temperature switches that input to the alarm were found out of calibration.

Also, the plant computer point for primary water after stator winding temperature was reading incorrectly.

The generator primary water system temperature is regulated by controlling the cold primary water flow. The post trip investigation revealed that the temperature controller was set at 113 degrees F. The generator hydrogen system temperature is regulated by controlling the cold hydrogen temperature leg at 104 degrees F. This results in a Delta T of only 9 degrees F (50 degrees C), which is less than the >10 degrees F specified in the log. The equipment was operating correctly, but the operating logs, the abnormal operating procedure, and the alarm procedure all required >10 degrees F Delta T; therefore, the procedures should have been different to reflect the established operating setpoints, or the setpoints should have been more conservative to correspond to the range specified in the operating procedures.

The Nuclear Regulatory Commission was informed of the event via the Emergency Notification System at 2337 per 10CFR50.72(b)(2)(ii).

E. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE OR PROCEDURAL ERROR

The reactor trip was annunciated by numerous alarms in the Control Room. The immediate cause of the event was identified by troubleshooting after the event.

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II. COMPONENT OR SYSTEM FAILURES

A. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT

The 'primary water flow temperature after main filter' temperature switch had failed and would not have actuated the generator primary water high temperature alarm.

The 'leak water temperature after cooler' temperature switch was out of calibration (>140 degrees F) and was reset to approximately 120 degrees F.

The 'primary water bushing outlet' temperature switch was out of calibration (200 degrees F) and was reset to approximately 176 degrees F.

The 'primary water rotor outlet' temperature switch was out of calibration (205 degrees F) and was reset to approximately 176 degrees F.

The plant computer point for primary water after stator winding temperature was reading incorrectly, the computer point was corrected. This point is not used as an alarm input and did not have an affect on this event.

None of these failures caused the event or were a result of the event.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

Not applicable, none of these failures caused the event or were a result of the event.

C. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF
COMPONENTS WITH MULTIPLE FUNCTIONS

Not applicable, the switch failure did not affect system operation or any secondary functions.

D. FAILED COMPONENT INFORMATION

Not applicable, the switch failure did not affect system operation or any secondary functions.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

The RPS and Auxiliary Feedwater System (EIIS:BA)) actuated during the event; all associated components within these systems functioned as designed.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

No safety system trains were inoperable as a result of this event.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

A turbine trip initiated by a generator trip leads to a reduction in the capability of the secondary system to remove heat generated in the reactor core. This event is analyzed in Section 15.2.3 of the CPSES Final Safety Analysis Report (FSAR). The analysis uses conservative assumptions to demonstrate the capability of pressure relieving devices and to demonstrate core protection margins. The event of January 8, 1992, occurred at 100% reactor power, and all systems and components functioned as designed. The event is completely bounded by the FSAR accident analysis which assumes an initial power level of 102% and conservative assumptions which reduce the capability of safety systems to mitigate the consequences of the transient. It is concluded that the event of January 8 did not adversely affect the safe operation of CPSES Unit 1 or the health and safety of the public.

IV. CAUSE OF THE EVENT

ROOT CAUSE

1. The BOP RO failed to understand the potential consequences to the generator primary water system by stroking closed the primary water temperature control valve. A formal briefing was not held prior to stroking the valve and the potential for this evolution to become a high risk activity was not considered.

2. The Shift Supervisor failed to ensure that the evolution was properly supervised nor did he communicate effectively to ensure that the evolution was proceeding correctly. When recommending that the control valve be checked for freedom of movement, the Shift Supervisor did not give the BOP RO specific instructions.

CONTRIBUTING FACTORS

1. The BOP RO failed to utilize available procedures. The BOP RO did not feel the use of the available alarm or abnormal operating procedures was necessary for the task.

2. The system operating procedure, abnormal operating procedure or alarm procedure would not have necessarily prevented the operator from taking the course of action that initiated this event. The specific actions taken in response to the Delta T problem were not contrary to these procedures. Even though it could have jeopardized the generator pressure drop test, the prudent action would have been to lower generator cold gas temperature rather than raise primary water temperature to restore Delta T.

3. Four inputs to the generator primary water high temperature alarm had failed or were out of calibration.

GENERIC CONSIDERATIONS

An evaluation should be made to determine the adequacy of the technical information available to operators to operate and maintain the turbine generator.

V. CORRECTIVE ACTIONS

CORRECTIVE ACTIONS TO PREVENT RECURRENCE

ROOT CAUSE

1. The BOP RO failed to understand the potential consequences to the generator primary water system by stroking closed the primary water temperature control valve.

CORRECTIVE ACTION

The BOP RO and his shift crew were given intensive training following the event. This event will also be covered in requalification training for all shift crews in the upcoming training cycle.

2. The Shift Supervisor failed to ensure that the evolution was properly supervised nor did he communicate effectively to ensure that the evolution was proceeding correctly.

CORRECTIVE ACTION

The Shift Supervisor was counselled after the event. The shift crew was also given intensive training on the factors leading to the event. Training on performing high risk activities and infrequent evolutions will be covered in the requalification training for all crews in the current and upcoming training cycles.

CONTRIBUTING FACTORS

1. The BOP RO failed to utilize available procedures. The BOP RO did not feel the use of the available alarm or abnormal operating procedures was necessary for the task.

CORRECTIVE ACTION

The BOP RO and his shift crew were given intensive training following the event which included the use of available procedures. This aspect of the event will also be covered in requalification training for all shift crews in the upcoming training cycle.

2. The system operating procedure, abnormal operating procedure or alarm procedure would not have necessarily prevented the operator from taking the course of action that initiated this event.

CORRECTIVE ACTION

The system operating procedure, abnormal operating procedure and alarm procedure will be revised to clearly define the preferred methods of manually controlling primary water temperature and the potential consequences of this activity. The recommendations of the task team will also be incorporated in these procedure changes as appropriate.

3. Four inputs to the generator primary water high temperature alarm had failed or were out of calibration,

CORRECTIVE ACTION

The four inputs were repaired and recalibrated. An investigation is being conducted to determine why these inputs failed or were out of calibration. The investigation will also examine the method used to calibrate these thermocouples.

GENERIC CONSIDERATIONS

An evaluation should be made to determine the adequacy of the technical information available to operators to operate and maintain the turbine generator.

CORRECTIVE ACTION

A task team has been formed and is evaluating the technical information available to operate and maintain the turbine generator. Task team recommendations will be reviewed and implemented by operations management as appropriate.

VI. PREVIOUS SIMILAR EVENTS

There have been no previous similar events reported pursuant to 10CFR50.73.

VII. ADDITIONAL INFORMATION

The times listed in the report are approximate and Central Standard Time.

ATTACHMENT 1 TO 9202120181 PAGE 1 OF 1

Log # TXX-92048
File # 10200
Ref. # 10CFR50.73(a)(2)(iv)
TUELECTRIC
February 7, 1992

William J. Cahill, Jr.
Group Vice President

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERED SAFETY FEATURE
LICENSEE EVENT REPORT 92-001-00

Gentlemen:

Enclosed is Licensee Event Report 92-001-00 for Comanche Peak Steam Electric Station Unit 1, "Reactor Trip/Turbine Trip on High Primary Water Temperature."

Sincerely,

William J. Cahill, Jr.

JET/tg

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (2)

P.O. Box 1002 Glen Rose, Texas 76043-1002

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